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PROGRESS IN RESEARCH ON CONTROLLED NUCLEAR FUSION

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The purpose of research on controlled nuclear fusion is to seek an ideal
new energy source for mankind. This is one of the most important items of science
and technology research of the whole world. Many countries spend enormous man-
power and material resources on this research. This paper briefly describes the
important progress in recent years in this field.

I. Progress in Research on a Magnetic Confinement System

In the magnetic confinement system, a magnetic field with special position
and shape confines the high-temperature plasma into a certain zone. There are
many types of experimental installation for the magnetic confinement. At pres-
ent, those making relatively faster progress are the Tokamak device, Stellarator,
magnetic mirrors, and reverse field pinch device.

(A) Tokamak device

This is a toroidal pole-less discharge type device. A current flows through
a toroidal vacuum chamber to heat and confine the plasma. The major characteris-

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tic of this type of device is that a longitudinal-direction strong magnetic field is used to successfully confine the macroscopic instability of the plasma. In the late 1960s, encouraging progress was made in the Tokamak device, which was stressed [1,2] by world countries as a result. A number of Tokamak devices were built in the 1970s. In recent years, several large Tokamak devices were built and operated with very good experimental results. Progress was mainly exhibited in the following aspects:

(1) Heating of plasma

There is a certain limitation on ohmic heating for Tokamak devices. With increasing temperatures, the specific resistance of plasma drops appreciably. In order to heat the plasma up to an "ignition" temperature, secondary heating means should be applied. As revealed by experimental results in recent years, the effective means of secondary heating are addition of neutron beams and high-frequency electromagnetic waves.

(a) Addition of neutron beam for heating: In this method, a high-energy neutron particle beam is added to the Tokamak plasma. By interaction, the energy of the high-energy neutron particles is transferred to the plasma. Experiments in using this method began in the early 1970s. At that time, only tens of kW of power were added; by now, the added power has been increased to several MW. In recent years, in the ORMAK and ISXB Tokamak devices at Oak Ridge in the United States, the DITE device in the United Kingdom, and the TFR device in France, experiments on adding neutron beams were conducted. In particular, in summer of 1978 in the largest operating Tokamak, the PLT device at Princeton in the United States, 4-channel 2.8 MW heavy current neutron beams were added for heating, with the ion temperature exceeding 60 million degrees [2,3]; the temperature was further increased to 75 million degrees, which is the best experimental result obtained up to the present time. It is really an encouraging result to those engaging in and concerned with the nuclear fusion research. In the United States, preparations were being made to conduct experiments of larger scale addition of neutron beams (with power up to 6 MW) at a large installation PDX with polar deflector. On a dual-stream device Doublet III, attempts were being contemplated to conduct a high power addition experiment on neutron beams [4].

(b) Ion cyclic resonance heating (ICRH): There are various types of oscillation in a plasma. If the added electromagnetic waves cause resonance with some particles in the plasma, this will obtain energy and result in higher temperature. In September 1979, at the Seventh European Controlled Nuclear Fusion and Plasma Physics International Conference held in the United Kingdom, the United States and French delegates announced the most recent experimental results [4] regarding ICRH. In the United States, heating experiments were conducted on a PLT device by use of high-frequency electromagnetic waves with 350 kW in power and 25 MHz in frequency. A single antenna wire was used to couple the electromagnetic waves with deuterium plasma containing traces of H^+ and $^3He^{++}$. These H^+ and $^3He^{++}$ ions obtain energy from the electromagnetic waves at an average of 10 keV. The distribution of energy is of the non-Maxwell type with a high-energy tail of 80 keV. After collisions occur among these high-energy particles, the temperature of the entire plasma can be increased. By use of this method, the ion temperature can be increased from 0.6 keV to 1.2 keV. For the next step, large scale ICRH heating using several antenna wires for simultaneous coupling with frequency at 43 MHz and power of several MW will be tested on the PLT device. In France, ICRH heating was also performed on a TFR device; the frequency used was 60 MHz and power at 250 kW. H^+ was also added into the deuterium plasma. In the experimental results, the ion temperature was increased to 150 eV with an average increase of 0.7 eV in ion temperature for each kW of power. In Japan, ICRH heating of 200 kW was applied on a DIVA device; as a result, the ion temperature was increased to 250 eV with an average increase of 2 eV in ion temperature for each kW of input power. In addition, the highest content of H^+ in the deuterium plasma was discovered to be 5-10 percent.

(c) Electron cyclic resonance heating (ECRH): Several years ago, in the USSR ECRH heating experiments were conducted on a TM-3 device. Recently, ECRH experiments [4] were also conducted on a large Tokamak device T-10. The high-frequency waves used had a frequency of 85 GHz (a secondary harmonic frequency) and power of 120 kW. As revealed by measurement results of soft X-rays of high-temperature plasma, the electron temperature was increased by about 200 eV.

(d) Heating with mixed waves: In Japan, low mixed wave heating experiments [4] were conducted on JFT-2 and JIPPT-2 devices and the power was 200 kW. As revealed by experimental results, the ion temperature was increased by 1 eV for each kW of input power.

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Recently some researchers proposed plans for heating by use of heavy current electron beam on the Tokamak plasma [5]. It was predicted that difficulty will occur in the addition technique.

(2) Increase of plasma density

For a certain range of longitudinal field B_ϕ and current I_p in a Tokamak device, there is an upper limit for plasma density. If the density is too high, breaking instability will occur on the plasma. For the present scale of Tokamak devices, generally the B_ϕ is about tens of kG, I_p is hundreds of kA, and the density of the plasma is generally $n \leqslant 5 \times 10^{13}$ cm^{-3}. However, in 1976 intermediate pulse gas filling was applied on a strong magnetic field Tokamak device Alcator-A in the United States; a high-density operation was carried out [6,7]. With a strong intensive magnetic field of 85 kG, the density exceeded 10^{15} cm^{-3}; confinement time $\tau_E = 20$ ms and $n\tau_E = 2 \times 10^{13}$ cm^{-3}·sec. It was discovered that the confinement time τ_E increases with increase of plasma density and the calibration relationship is $\tau_E = 1.7 q^{1/2} \bar{n}_{14}$ ms; in the equation, q is the safety factor

$$\bar{n}_{14} = \frac{n}{10^{14} \text{cm}^{-3}}.$$

Thus, $n\tau_E \propto n^2$ is derived. This explains that the increase of density is very advantageous to attain the ignition state. This is the best parameter obtained up to now and likely to give encouragement to those concerned with and engaging in nuclear fusion research. We noted that for the PLT device, $T_i = 6$ keV; for the Alcator-A device, $n\tau_E = 2 \times 10^{13}$ cm^{-3}·sec. These are important milestones in the progress of nuclear fusion research in recent years. This indicates that the target of "ignition" is not far away. Of course, these two conditions should be simultaneously met at the same device.

(3) Low q steady operation

In order to obtain steady plasma position and shape, the Kruskal-Shafulanoufu [transliteration of Russian name] stability criterion should be satisfied:

$$q = \frac{B_\phi}{B_\theta} \cdot \frac{a}{R} > \frac{2}{2 - \beta},$$

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In the equation, B_ϕ is the longitudinal magnetic field; B_θ is the angular magnetic field; a is the radius of the plasma; R is the radius of the large toroidal discharge chamber; β is the ratio of the pressure intensity of the plasma and the intensity of the magnetic potential $B^2/8\pi$; and q is called the safety factor. In order to have a stable macroscopic position and shape of the Tokamak plasma, generally q≳2.5∿4. For the device, a and R are constants. In order to ensure the value of q within the safety range, the plasma current and the longitudinal field B_ϕ should be increased; this will lead to greater cost. In recent years, studies on the minimum value of the safety factor q for the Tokamak device were conducted. The study undoutedly is advantageous in saving costs, for a more economical Tokamak reactor. Recently, studies were made on low q operation on a large Tokamak device T-10, which was completed and operated in USSR. As indicated by experimental results [4], when q(a)=1.6 the discharge is still steady. During low q discharge, the light impurities are predominant; approximately 50 percent of the input power is radiated away. For the T-11 device, low q discharge experiments were also conducted with an effective atomic number Z_{eff}=1. As indicated by experimental results, when q(a)=1.2-1.4, confinement of plasma is very poor. For a DIVA Tokamak device in Japan, very low q discharge experiments [4] were also conducted; the method applied was the combination of rising plasma current with gas filling. The minimum value of q(a) is 0.85. When q(a)<2, no appreciable breaking of plasma was observed. However, when q(a)<1.3, confinement of plasma as a whole was very poor. The results are consistent with that of the T-11 device.

(4) Operations of some newly built (or remodeled) large Tokamak devices [4]

At present, a large PDX Tokamak device with a polar deflector has begun operation in the United States. The discharge parameters have attained the related values; the longitudinal field B_ϕ=25 kG; plasma current I_p=500 kA; the average electron density $\bar{n}_e$=2x10^{13} cm^{-3}; the electron temperature $T_e(0)$=1.4 keV; the ion temperature $T_i(0)$=0.6 keV; and the energy confinement time τ_E=30 ms. Later, deflectors will be added and heating with the addition of neutron beams (6 MW) will be used. It is predicted that better parameters will be obtained at that time. Another large new device, Doublet III, of the General Atomic Corporation in the United States was completed; its main purpose is to study the steady position and shape of non-circular cross-section plasma with the following parameters:

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R=140 cm, a=90 cm, b=270 cm (a and b are two axes similar to the elliptical cross section), longitudinal field B_ϕ=26 kG, the plasma current attained is 1.5 MA, electron temperature $T_e(0)$=1 keV, $q(0) \gtrsim 1$, and the energy confinement time τ_E=10 $\sim$ 30 ms. Another new device recently operated is the Alcator-C, larger than the Alcator-A, which successfully operated with high density plasma. The design parameters are as follows: R=64 cm, a=17 cm, $B_\phi < 120$ kG, $I_p \leq 1$ MA. At present, the attained data are B_ϕ=60 kG and I_p=500 kA. As discovered from the preliminary experimental results, the energy confinement time τ_E has not increased with increasing density. From analysis, it is considered that this result may be because the current operation of the Alcator-C has not attained the most desirable state and the discharge is not sufficiently clean.

In addition, there are several larger Tokamak devices (large radius R is about 3 m and small radius r, 1$\sim$2 m); such devices include the TFTR in the United States, the JT-60 in Japan, and JET in Europe (in joint operation by several countries). These devices are under construction.

2. Stellarator device

The Stellarator is a toroid-shaped device experimented with and studied since the early 1950s. The magnetic field has a spiral transformation with low β operation similar to the Tokamak device. The main distinction between the two is that the spiral transformation magnetic field in the stellarator is produced by spiral winding outside the plasma container. In the Tokamak, the magnetic field is the superimposition of the external applied longitudinal strong magnetic field and the angular magnetic field produced by the plasma current. Since the experimental results of stellarators were poor in the 1950s and 1960s, progress was slow. At one time, the stellarators were not highly regarded. In the United States, basically studies of stellarators have been suspended since the late 1960s. However, studies in the United Kingdom, West Germany, Soviet Union and Japan were continued. In the mid-1970s, better experimental results were obtained on large stellarators, such as the CLEO, WVII and L-2 devices; these devices use ohmic heating in their operation. Under conditions of 6$\sim$35 kA current and 12$\sim$35 kG magnetic field, the attained results are as follows: electron temperature T_e=200$\sim$900 eV; ion temperature T_i=100$\sim$500 eV; density n=5x10^{12} $\sim$ 6x10^{13} cm^{-3}; and the energy confinement time τ_E=1$\sim$10 ms. Because of these achieve-

ments and the potential advantages of steady operation, these devices were again highly regarded by researchers. In September 1979 at the Ninth European Controlled Nuclear Fusion and Plasma Physics Conference convened in the United Kingdom, delegates from the USSR, United Kingdom and West Germany reported the most recent research results [4] of stellarators.

In recent years, the Karam Laboratory in the United Kingdom conducted low-current ohmic heating experiments; the researchers discovered that the confinement of plasma was quite good under these situations without hydromagnetic instability. The typical parameters are as follows: B_ϕ=18 kG, I_p=7 kA, $\bar{n}_e$= 2×10^{13} cm^{-3}, T_e=140 eV, and T_i=150 eV. The corresponding energy confinement time is 5 ms and the ohmic heating input power is 12-15 kW.

The WVIIA device in West Germany is a large stellarator: the large radius R is 200 cm; the spiral winding ℓ=2; and the changeable rotating angle can be adjusted continuously. When the magnetic field is 40 kG, the range of the changeable rotating angle is 0~0.23. At 20 kG, the range of the changeable rotating angle is 0~1. The WVIIA device applies the method of high frequency pre-ionization to obtain plasma. From the experimental results of the WVIIA and CLEO stellarators, it was discovered that the distributions of temperature and density of plasma are much flatter than that obtained in Tokamak devices. This indicates that stellarators can have higher energy densities than the similar scale Tokamak devices. Recently, experiments were performed with the changeable rotating angle 0.14, I_p=20 kA, B_ϕ=35 kG; as a result the energy confinement time began to increase with increase in density n like that in the Alcator scale rule. However, the confinement time began to be reduced later. It was also discovered in experiments that when the confinement time of the CLEO device began to decrease, the plasma density corresponded to the density of Tokamak plasma when beginning to break up. On the CLEO stellarator, successful experiments on neutron beam addition were performed. The addition power was 450 kW; particle energy was 30 keV, and absorption efficiency was 30 percent without causing an increase in impurities. It was discovered that the addition effect is better when the plasma current is smaller. During low current operation, internal breaking was avoided when m=2. In the plan, the addition power will be raised to 1 MW.

At present, a number of large stellarators are under construction. At Kyoto University in Japan, a large Helitron-E device with large radius R=220 cm is under construction. The toroidal magnetic field (produced by the spiral winding) B_ϕ=20 kG. In Australia, a component-type large stellarator is under construction; the device is composed of eight independent components with spiral winding ℓ=3. The large radius R=200 cm; small radius a=20 cm; and the toroidal magnetic field B_ϕ=40 kG.

3. Installation of magnetic mirrors

The installation of magnetic mirrors is an open-end system, which uses a magnetic field position and shape (weak at the middle and strong at both ends) to confine plasma. There are advantages of the system as follows: simple structure, high β value, capable of steady operation and relatively convenient in installation and maintenance. Therefore, if in the future a magnetic mirror type fusion reactor is built, the reactor may be relatively economical and practical. The main disadvantages are open magnetic lines in the installation, resulting in relatively heavy loss of particles at the ends of the device. In recent years, important progress was obtained in stability and confinement of plasma. In 1976 at the 2XIIB magnetic mirror installation at Lawrence Livermore Laboratory in the United States, the ion temperature attained 13 keV and the density, 2×10^{14} cm^{-3}. Moreover, during addition of a neutron stream, no instability of plasma is induced. Therefore, for the magnetic mirror device the main problem requiring a quick solution is trying to reduce the terminal losses and to increase the confinement time of plasma. Therefore, consideration should be made of two aspects: on one hand, recycling the energy lost from the terminals should be conducted. On the other hand, a complex shape of magnetic field should form a "plug" to stop leakage. Based on these ideas, new concepts of reverse field magnetic mirrors, series magnetic mirrors and toroidal chain magnetic mirrors were advanced in recent years.

The United States, Soviet Union, United Kingdom, France and Japan engaged in research on installation of magnetic mirrors; especially the United States contributed the most resources in the research, developing installations of reverse field magnetic mirrors and series magnetic mirrors, which are the most

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stressed. The reverse field magnetic mirror applies some means to close the magnetic lines of these mirrors, so that the plasma cannot escape along the magnetic lines through the terminals. In the 2XIIB device, a method of neutron stream addition to reverse the magnetic field was used; success was more or less achieved. In a new experimental plan, low temperature plasma will be used to reverse the magnetic field. The so-called series magnetic mirror device connects two short terminal magnetic mirrors at both ends of a long intermediate magnetic-mirror chamber (or spiral tube). High-temperature, high-density "plugging" plasma is thus formed at the terminal magnetic mirrors using electrostatic confinement to stop the terminal loss. At Lawrence Livermore Laboratory in the United States, large series magnetic mirror devices have been operated [4]. The intermediate spiral tube is 5 m long; the core magnetic field is 2 kG; at both terminals, there are two terminal magnetic mirrors with field intensity of 10 kG, and each terminal mirror is heated by addition of a 5 MW neutron beam. At present, the added power is about one third of the rated value. The plasma density in the spiral tube is 10^{13} cm^{-3} and the plasma temperature is 100 eV. At the terminal magnetic mirrors, the average energy is 10 keV and the persistent time of the heated beam stream is 25 ms. An encouraging phenomenon in experiments is the reduction of losses caused by electron thermal conduction. Viewed from the development trend, the series magnetic mirror device may be a magnetic confinement device with good prospects of competing with the Tokamak device.

4. Reversed field pinch device

The reversed field pinch device was accidentally discovered during a discharge experiment of the Zeta device at the Karam Laboratory (in 1965) in the United Kingdom. Later, a method of program-controlled external circuit current was used to establish the position and shape of this type of magnetic field. The characteristic of the device is that the toroidal magnetic field B_ϕ reverses its direction near the outer surface of the plasma cylinder; therefore, the magnetic shearing is large, capable of attaining stability equilibrium in the condition $q<1$. A number of new reversed field pinch devices are undergoing construction. Among them, the ETA-BETA II device was completed and operated [4] in 1979. The large radius of the device is 130 cm; the small radius is 25 cm; the toroidal magnetic field is 3 kG as measured in experiments; the toroidal current is 210 kA;

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and the attenuation time of the current is 1.5 ms. When the reversed field
pinch type plasma is formed, temperature increases from 40 eV to nearly 100 eV;
the fluctuation of dI/dt is appreciably reduced; and the confinement is also
enhanced. The reversed field pinch device is also one of the magnetic confine-
ment devices that is highly regarded at present.

II. Research and Progress on the Inertial Confinement System

The basic principle of inertial confinement is the use of some means to
suddenly heat traces of deuterium and tritium fusion materials to a required high
temperature so that due to inertia the matter is unable to scatter in time for the
fusion reaction to take place. At present, the explored and studied inertial
confinement systems include laser nuclear fusion, relativistic intensive electron
beam nuclear fusion, and intensive current ion beam nuclear fusion (including
light ion beams and heavy ion beams). If we compare the inertial confinement
system with the magnetic confinement system, the inertial system avoids the
complex heavy magnetic field configuration; therefore, the inertial system is
quite attractive. Although the research task was performed at a later time,
considerable progress was achieved in the span of more than a decade. The inertial
confinement system has been more and more emphasized.

1. Laser nuclear fusion

In the 1960s, laser technology was rapidly developed to open a new field
(laser nuclear fusion) of nuclear fusion research. In the early 1970s, larger
scale laser nuclear fusion research was developed, with considerable progress
being achieved in recent years. The United States contributed considerable
resources to the research; in the Lawrence Livermore Laboratory, large power
laser systems were mainly studied, using neodymium glass as the working medium.
In 1977, an Argus device was built with output power of 3×10^{12} W, 2 kJ in
energy, the ion temperatured was $T_i = 8$ keV and the neutron output was 10^9. In
1978, a large-scale laser nuclear fusion device, SHIVA [4], was completed. The
device is 6 m high and 90 m long; there are 20-channel laser beams, 2.6×10^{13} W
in power, and output energy of 10^4 J. By use of such gigantic multiple channel
laser beams for illumination of deuterium-tritium target balls, the observed

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compression of liquid density was about 100 times and the neutron output was 3×10^{10}. This is the highest level achieved in current laser nuclear fusion research. Those researchers planned to complete in 1982 a larger-scale neodymium glass laser nuclear fusion device, NOVA [4], with a power of 2×10^{14} W and output energy of 100 kJ. In the Los Alamos Laboratory, the laser nuclear fusion research uses CO_2 as the working medium; the laser power approaches the level attained by the neodymium glass laser.

In the USSR, there are two large laser nuclear fusion devices: Kalmar and Delfin [4]. There are nine laser channels in the Kalmar device with energy of 200 J. The Delfin device is quite large in scale, with 216 laser channels: the total energy is 10 kJ and the pulse width is 0.1∿10 ns. At present, the operating Delfin device has 54 channels of laser beams with energy of 500 J and pulse width 1.5∿5 ns. In Japan, a large laser nuclear fusion device is also under construction.

2. Relativistic intensive current electron beam nuclear fusion

This is a new field with relatively late development in nuclear fusion research; its basic principle is similar to laser nuclear fusion. Comparing the two approaches, the advantage of this approach is the high efficiency of converting electric energy into electron beam energy. Technically the building of the high power electron beam device is relatively simple; therefore this approach has been more and more stressed. Of the disadvantages, with the exception of relatively difficult focusing and transmission of electron beams, the major difficulty occurs during bombardment of electron beams. The target pill is preheated by radiation; thus, it is difficult to compress. According to the classical Coulomb collision theory, the collision probability is very small between the relativistic electron beams and the electrons and ions in the target. The encouraging aspect is that these experiments have indicated that the energy deposit due to interaction between the heavy current relativistic electron beams and the target particles is greater by 1∿2 magnitudes than the calculated value in the classical theory. The relativistic electron beams are mainly produced through a pulse electron accelerator. In the 1970s, several large pulse electron accelerators were built in the United States, Soviet Union and Japan. At present, the largest operating device is the Proto 2 device in the United States: the

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voltage is 1.5 MV; current is 6000 kA; pulse width is 24 ns; total energy is 216 kJ; and the power is 9×10^{12} W. Several larger heavy current electron-beam-producing devices were undergoing construction [8]. At Sandia Laboratory in the United States, two high-power electron beam producing devices are undergoing construction; they are the EBFA-1 and EBFA-2 devices. The EBFA-1 device is planned to be completed in 1980: the voltage will be 2 MV and the net energy transmitted to the load will be 1 MJ. The device is composed of 36 components; the first component underwent experiments in 1979. The EBFA-2 device will be completed in 1983-1984 with a design power of 10^{14} W. In the Soviet Union, a plan was underway to build a larger heavy current electron beam producer, the Angara-5 device, at the Kurchatov Institute. The designed energy is 5 MJ and it is planned that the device will be completed in 1983-1984.

3. Heavy ion beam nuclear fusion

This is a new scheme proposed in recent years by using heavy ion beams to compress and heat the target ball for nuclear fusion. The basic principle is that microsecond magnitude high energy heavy ion beams are led out of a heavy ion accelerator to separately enter into storage toroids. Then, from the storage toroids the nanosecond pulse width heavy ion beams are simultaneously led out to hit the target ball for compression and heating. Advantages of this scheme are that the scattering of ions is small on the outer shell of the target ball; there is no braking radiation and the energy deposit is high; therefore, the required total power of the ion beam, relatively speaking, is smaller. The disadvantage lies in difficulties of leading out and transmission of ion beams, resulting in high construction cost. It is estimated that with current accelerator technology it is possible to produce heavy ion beams suitable for nuclear fusion but the cost is relatively higher. At the Argon Laboratory in the United States, it is planned to conduct in 1984 demonstration experiments [8] of heavy ion beam nuclear fusion. On this demonstration basis, study then can be started for experiments corresponding to the gain and/or loss of energy.

4. Bombardment of light ion beams onto target

At a session of the Heavy Current Electron Beam and Heavy Current Ion Beam Third International Special Topic Discussion Meeting convened in July 1979, the

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method of using light ion bombardment for carrying out inertial confinement [8] was highly regarded. On one hand, this is because apparent progress has been made in producing and focusing heavy current ion beams. On the other hand, the energy absorption efficiency between the beam current and the target in using this scheme is much higher than that with laser nuclear fusion. The US Navy Science Laboratory and the Sandia Laboratory reported their research progress. At the focus of light beams, the current density has attained as high as 200 kA cm^{-2}. They are in the process of improving focusing technology, to enable the current density at the target to reach $10^7 A\ cm^{-2}$. At the Limeil Laboratory in France, heavy current deuterium beams of 1.2 MA current intenisyt and 300 keV energy were achieved. These data are quite attractive for inertial confinement fusion; however, the difficulty is the relatively large divergence angle of the beam current.

III. Research on Fusion Reactor

Due to constant encouraging experimental results achieved in various types of nuclear fusion devices, as well as progress in high-temperature plasma physics, researchers' confidence in building a fusion reactor for controlled nuclear fusion power has increased. By the year 1970, research on fusion reactors were begun. The manpower and resources contributed on engineering and technology of the fusion reactor were increased by the various countries year after year; several times international fusion reactor technology conferences were convened. At present, research on the fusion reactor is basically proceeding in two aspects. On one hand, based on several promising major experimental devices, concept design on fusion reactor was performed. Several concept design schemes of different types of systems were pursued, including Tokamak fusion reactor, magnetic mirror fusion reactor, and laser fusion reactor. The scheme includes the system design of the reactor, starting, control and addition of fuel to the reactor, as well as removal of spent fuel and scraps. On the other hand, such problems are dealt with as radiation damage of th first wall material, control of impurities, and the breeding cycle of deuterium. Considerable work has been done on these aspects. At present, an engineering test device (ETF) has become the central task of the nuclear fusion venture in the United States. ETF is a Tokamak type ignition device [4] using deuterium and tritium as fuel; the large radius (of ETF) is R=5 m and the small radius is (1.2/1.9) m. The $<\beta>\sim 7$ percent; the combustion time is greater than 100 seconds; and the heat load of the first wall is greater than 1 MW/m^2. In December 1979, under the sponsorship of the International Atomic Energy Commission, four delegations, from the United States, Soviet

Union, Japan and Western European Joint Organization, jointly began to design a large international fusion device (INTOR) with large radius of 5.2 m and small radius of 1.6 m.

IV. Conclusions

Controlled nuclear fusion research is a major attack of mankind's exploitation of nature. However, a series of difficult problems have to be solved (theory, experiments and engineering technology) before attaining the target of nuclear fusion power. However, up to now there has discovered no unsurmountable barrier to the progress route. According to estimates [9] by the international fusion research organizations, in the 1980s, it will be possible to achieve thermal nuclear "ignition". In the early years after 2000, it is a high hope to use nuclear fusion power to supplement the energy shortage expected to begin in these years. Beginning in the 2020s, nuclear fusion power will occupy a more and more important position in the global sources of energy.

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FAST HYDROGEN PURIFICATION BY USING HYDROGEN PROPELLING DEVICE

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During normal operation, a hydrogen propelling device requires very high hydrogen purity; even with traces of impurities the performance of the device will be seriously affected. The oscillation of the propelling device will be suspended [1] if O_2 in the storage bubbles has a partial pressure as low as 10^{-7} atmospheric pressure. Generally, industrial hydrogen contains large amounts of impurities, such as water vapor, CO, CO_2, O_2 and N_2; therefore, only after purification can such industrial hydrogen then be used in a propelling device. The process of chemical purification is complex and the end product has a low purity. This paper introduces a purification method using a palladium-silver tube; the method is simple but the purity of the end product is high, over 99.99 percent [2]. The flow of the hydrogen stream can be steadily controlled and its flow rate can be quickly changed. Therefore, besides being used in a hydrogen propelling device, the purified hydrogen can be utilized in nuclear physics experiments, hydrogen stream technique, and hydrogen-filling devices.

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I. Simple Working Principle of Purifier

A method of obtaining highly purified hydrogen is to use a thin metal wall (such as palladium, nickel or iron) as a screener to remove impurities. When the thin metal wall is heated, hydrogen can infiltrate through it but other gases either infiltrate in trace amounts or not at all. A single volume of palladium can dissolve hundreds of volumes of hydrogen; therefore, the infiltration rate of hydrogen to palladium is the highest. However, there are two difficulties in using a pure palladium tube: (1) with a temperature higher than 300°C, very quickly a coarse structure (of crystal lattice) will form, possibly causing leakage of gas through cracks; and (2) palladium can only be welded with gold; welding with other metals is quite difficult if at all possible [3]. In 1960, Hunter purified hydrogen by use of a palladium-silver alloy diffusion method, since the alloy will not lead to crack leaking of gas (or form a coarse crystal lattice structure) when the alloy is placed in hydrogen gas whether it is cold or hot. At 500°C, the infiltration rate of hydrogen is 25 percent higher than the rate of pure palladium; at 300°C, the infiltration rate of hydrogen is 150 percent higher than the rate of pure palladium (in this case, the oxyten content is lower than 10^{-9}) [4].

The expression equation of infiltration rate K of hydrogen to palladium is

$$K = K_0 p^{1/2} e^{-C/T}.$$

K is the number of milliliters of gas passing through each millimeter thick of wall per second for each square centimeter of surface while the pressure difference of two sides is 10 atmospheric pressures. $K_0 = 4.31 \times 10^{-3} S_0$; S_0 is the number of milliliters of gas dissolved by 100 grams of palladium under standard conditions; C is a constant relating to heat absorption; T is the absolute temperature K of palladium; and p is the pressure of gas with the atmospheric pressure as a unit. We can see from the above equation that the infiltration amount of hydrogen through palladium is related to surface area and thickness of the palladium tube wall, as well as the pressure and temperature of the gas. The effect of temperature is predominant. At the beginning, the amount of infiltration quickly

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increases with increase of temperature until a saturation value is approached. In a given palladium tube, only by controlling the temperature of the palladium tube while maintaining other conditions constant, the flow of the hydrogen stream after purification can be controlled.

II. Control of Hydrogen Flow and Fast Purifier

By use of an alloy with 75 percent Pd and 25 percent Ag to make a tube $\phi 3 \times 0.2$ square millimeters and about 30 mm long, the alloy tube is welded in a stainless steel tube, whose outer surface is wound with resistance wire in forming an intermittent heating type hydrogen purifier. Figure 1 shows a schematic diagram of the hydrogen flow control system. High pressure industrial hydrogen enters a purifier which encloses a heated palladium-silver tube. The current intensity of resistance wire is controlled by a flow servo circuit. A pressure gauge piping is connected to a hydrogen flow piping as the flow is to be controlled. The pressure gauge and a resistance vacuum gauge convert variation of hydrogen pressure into electric signals, which are fed into the servo circuit in order to control the heating current of the palladium-silver tube, as well as to control the hydrogen flow after its purification.
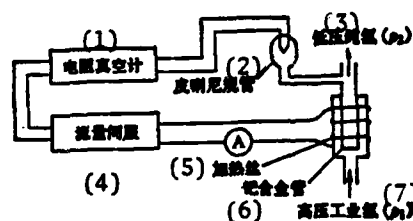

Fig. 1. Schematic diagram of hydrogen flow servo system.
Key: (1) Resistance vacuum gauge; (2) Pitot tube; (3) Low pressure pure hydrogen; (4) Flow servomechanism; (5) Resistance wire; (6) Palladium alloy tube; (7) High-pressure industrial hydrogen.

In the intermittent-heating type purifier, however, there is the phenomenon of heat lag since a period of time is required for heat transmitted from the resistance wire to the stainless steel tube and then transmitted to the palladium-silver tube. The hydrogen flow cannot be quickly adjusted. A direct heating type fast purifier does not have such a shortcoming on flow adjustment; the structural principle is shown in Fig. 2. A cylindrical tube with one end closed is made of the same palladium-silver alloy (as mentioned above), 40-75 mm long and $\phi2$x0.1 square millimeter. The cylindrical tube is then welded in a stainless steel tube. The resistance of the palladium-silver tube is about 20-40 milliohms. While a heating current of several amperes (controlled by a servo circuit) passes through the alloy tube, it heats up and its temperature increases due to the joule heating effect. Thus, the time required to transmit heat is shortened without the phenomenon of heat lag. The response time of the palladium tube from passing of the current for heating to the infiltration of hydrogen is about 4 seconds. The hydrogen flow can be quickly adjusted.
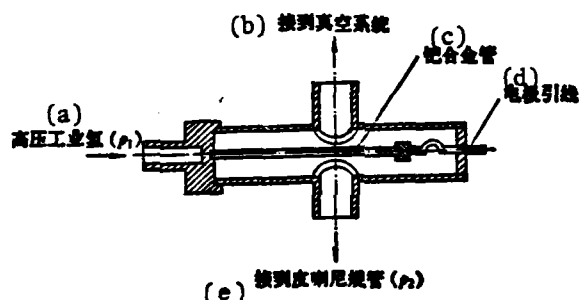

Fig. 2. Assembly principle diagram of fast hydrogen purifier.
Key: (a) High pressure industrial hydrogen; (b) Connected to a vacuum system; (c) Palladium alloy tube; (d) Electrode lead wire; (e) Connected to a pitot tube.

III. Experimental Results

Several direct-heating type purifiers were made with the same diameter and wall thickness of palladium-silver tubes but different lengths; the resistance

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values were in the range of 20 to 40 milliohms. The tubes were then heated and the relationship between the heating direct current and the hydrogen pressure in the gauge tube was plotted into curves, which represent the relationship between the temperature of the alloy tube and the infiltration amount of hydrogen. Figure 3 shows curves of two different input hydrogen pressures for three purifiers of different resistance values (indicating different lengths of palladium-silver tubes). Table 1 lists resistance values of three purifiers and two values of hydrogen pressures. We can see from Fig. 3 and Table 1 that the infiltration amount of hydrogen is greater when the input hydrogen pressure (to the same alloy tube) is higher. For longer alloy tubes with the same internal and external diameters, the infiltration amount of hydrogen is greater. Most important is the effect of the infiltration amount of hydrogen by the heating current, which determines the temperature of the palladium-silver tube.
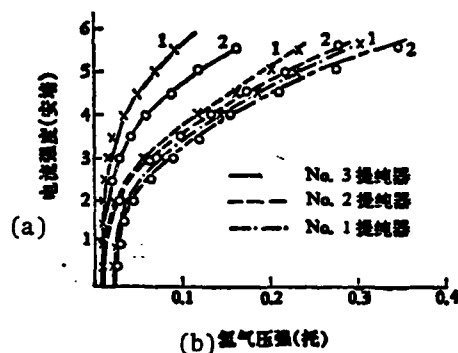

Fig. 3. Relationship between heating current (to palladium-silver tube) and hydrogen pressure measured by pressure gauge: ___ No. 3 purifier; - - - No. 2 purifier; -·-·- No. 1 purifier.
Key: (a) Current intensity (amperes); (b) Hydrogen pressure (in atmospheric pressure).

A hydrogen stream from the purifier is led to ionized bubbles, which have been sucked into a pressure of lower than 10^{-3} atmospheric pressure. When the

hydrogen pressure in bubbles lies with the range between 5×10^{-2} and 0.5 atmospheric pressure, the gas is excited with a radio frequency electric field to ionize the hydrogen molecules into atoms while exhibiting beautiful rosy red light. While in hydrogen gas without purification or the pre-sucked pressure in the bubbles is still higher than 10^{-3} atmospheric pressure, impurities in the bubbles still occupy some percentage; milky-white or pink light will be exhibited after it is excited with a radio frequency electric field. In these cases, the ionized hydrogen atom stream is again led into storage bubbles through a focusing system. Only when rosy red light is exhibited from the ionized bubbles, will the hydrogen propelling device oscillate. From the color of the light exhibited from the ionized bubbles and oscillation of the hydrogen propelling device, these phenomena indicate the fact that the purity of the hydrogen stream after purification is above 99.99 percent [1,2].

The time required for the hydrogen stream attaining an equilibrium state is related to the range of flow variation. The time required is longer for a larger variation range and shorter for smaller variation. Within the range required to satisfy automatic tuning of the propelling device, the required time is about 30 seconds for the whole system to change flow as shown in Fig. 4. The time required is as long as 20 minutes for the intermittent heating type.

Table 1. Different values of resistance and different input hydrogen pressures in a purifier.

(a) 提纯器编号	(b) 电阻值 (毫欧)	(c) 输入氢压(大气压)	
		(d) 曲线 1	(e) 曲线 2
No. 1	23	2	4
No. 2	25	6	7.5
No. 3	22	2.5	5

Key: (a) Number of purifier; (b) Resistance value (in milliohms); (c) Input hydrogen pressure (in atmospheric pressures); (d) Curve 1; (e) Curve 2.
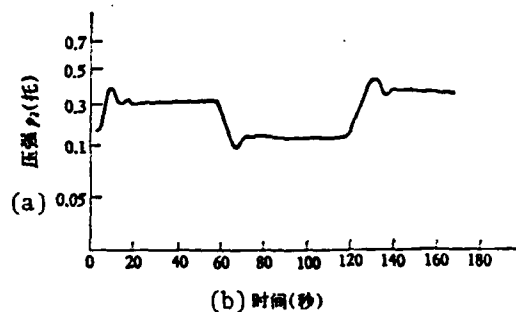
Fig. 4. Time required for changing flow to a
required value.
Key: (a) Pressure intensity p_2 (in atmo-
spheric pressure); (b) Time (seconds).

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